VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

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U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
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Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2
SURRY POWER STATION UNITS 1 AND 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
DOMINION'S RELOAD NUCLEAR DESIGN METHODOLOGY TOPICAL REPORT

Dominion's Reload Nuclear Design Methodology Topical Report has been revised to support the transition to Framatome ANP Advanced Mark-BW fuel at North Anna. In a letter dated October 8, 2001 (Serial No. 01-628) Dominion submitted Revision 2 of VEP-FRD-42, "Reload Nuclear Design Methodology Topical Report," for NRC review and approval. During review of the topical report, the NRC staff identified additional information that is needed to complete their review. This additional information is provided in the attachment to this letter.

If you have any further questions or require additional information, please contact us.

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachment

Commitments made in this letter: None

A001

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Attachment

REQUEST FOR ADDITIONAL INFORMATION
DOMINION'S RELOAD NUCLEAR DESIGN METHODOLOGY TOPICAL REPORT
VEP-FRD-42, Revision 2

North Anna Power Station Units 1 and 2
Surry Power Station Units 1 and 2
Virginia Electric and Power Company
(Dominion)

In April 15 and 16, 2002 discussions with the NRC staff, regarding Dominion's Topical Report, VEP-FRD-42, Revision 2, "Reload Nuclear Design Methodology," the following additional information was requested.

Question 1:

Is the Dominion reload methodology discussed in Topical Report VEP-FRD-42, Revision 2, intended to be applicable only for Westinghouse and Framatome ANP fuel types? If the intent is for other fuel types, please provide a discussion regarding how applicability determinations will be made and the process for determining the need for prior NRC approval.

Response:

The methodology discussed in VEP-FRD-42, Revision 2 is supported by extensive nuclear design predictions that encompass various evolutionary changes in fuel design features for Westinghouse fuel. Such predictions have been made for more than 40 reload cores, loaded in both North Anna and Surry reactors. Although the intended extension of this methodology is for the analysis of Framatome ANP fuel, the methodology is sufficiently robust for use on any fuel product with similar features. The methodology has several key elements, none of which are inherently dependent upon a specific fuel design or manufacturer. These key attributes of the methodology are:

- Analysis framework in which safety analyses establish the acceptable values for reload core key parameters, while nuclear and fuel design codes confirm each core's margin to the limits

- Use of bounding key parameter values in reference safety analyses

- Recurrent validation of nuclear design analytical predictions through comparison with reload core measurement data

- Representation of key fuel features via detailed inputs in core design and safety analysis models

- Fuel is modeled using approved critical heat flux (CHF) correlations demonstrated to be applicable and within the range of qualification

The Dominion reload design methodology focuses upon determining appropriately conservative values for two types of parameters: 1) the bounding value for key parameters assumed in the safety analyses and 2) the values for these same key parameters calculated for each reload core. The first parameter set constitutes the allowable limits for which the existing safety analyses remain valid. The reload values are determined for each specific core with the objective of confirming that they remain within the limit values. Application of this methodology to alternate fuel types would be accomplished in a fashion that preserves this fundamental approach. Prior to the use of the Dominion nuclear reload methodology for other fuel types, it is necessary to confirm that the impact of the fuel design and its specific features can be adequately modeled with the Dominion nuclear design and safety analysis codes. This includes comparison with appropriate benchmark data to confirm the capability to model the specific fuel features and to determine the inherent accuracy of such predictions. Results of these comparisons would also be used to determine whether any changes are needed in uncertainties that are applied to the nuclear calculations. If the features of an alternate fuel design can be modeled with comparable accuracy to the existing models and fuel design and require no change in the applied uncertainty factors, the applicability of the nuclear design portion of the methodology is established. This approach confirms that there should be no significant effect upon calculated values of reload key parameters. To determine applicability of safety analysis codes for analysis of alternate fuel products, a similar modeling capability assessment would be performed. This assessment would involve incorporating the appropriate detailed fuel design inputs into safety analysis code calculations and verifying that existing codes and models conservatively model the fuel behavior. This would be accomplished either by direct evaluation of the key phenomena or comparison to available vendor calculation results. The need to obtain prior NRC approval for these changes is governed by the requirements of 10 CFR 50.59, which in Sections (a)(2) and (c)(2)(viii) includes provisions that are relevant to methodology changes. If the changes necessary to accommodate another fuel product required changes to the reload methodology of VEP-FRD-42, Revision 2, these would be submitted for prior NRC review and approval.

Question 2:

The licensee states that the minor changes in Framatome ANP fuel features that could affect safety analysis design inputs are within the modeling capability of Dominion safety and core design analysis codes. Please verify that Framatome ANP fuel features are within all restrictions and limitations of Dominion safety and core design analysis codes.

Response:

Core Design Models

From a core design perspective, the differences in modeling Framatome ANP fuel relative to Westinghouse fuel are small and are accommodated using model input parameters. These differences are similar in magnitude to incremental changes in Westinghouse fuel over time, which have been successfully modeled. Minor changes include spacer grid differences, a slight increase in fuel density, and a slight difference in the position of the fuel stack. The grid differences are primarily due to the presence of intermediate flow mixer grids. In the PDQ and NOMAD models, grids are not explicitly modeled, but are homogenized over the entire length of the fuel stack. The effect of more grid material (primarily zirconium) is directly modeled in PDQ via input parameters (treated as nuclides) representing grid material and moderator displacement. The macroscopic cross section effect is transferred to the NOMAD model from PDQ. Similarly, cross sections in the PDQ model are a function of fresh fuel isotopic content; therefore, the density effects are also directly modeled.

Minor changes in fuel alignment have occurred in the past due to evolutionary changes in Westinghouse fuel products, such as the incorporation of protective lower grids. If there is a significant shift in the relative alignment of the burnable poison (BP) and the fuel, the burnable poison position is directly modeled by axially volume weighting the BP input in the axial nodes where the BP/fuel boundary changes. Comparison of measured and predicted Framatome ANP lead test assembly (LTA) axial and integral power distributions over three cycles of operation provides direct confirmation of the accuracy of the axial weighting, grid modeling, and fuel density modeling techniques.

RETRAN Models

In preparation for application of the Dominion RETRAN model to Framatome ANP fuel, specific card (record) overlays to the RETRAN input cards were developed. These overlays were developed such that appending them to the end of the current, Westinghouse fuel based model creates a Framatome ANP-specific RETRAN model.

<u>Fuel properties</u>

The Framatome ANP overlays were developed from fuel and clad properties data supplied by Framatome ANP which are consistent with those used in the approved Framatome ANP safety analysis models. Formal documents developed under the Framatome QA program were developed to transmit this data. Fuel properties covered included:

- Material properties of the three conductor materials (the fuel pellet, the pellet-cladding helium gap, and the M5 cladding)
  - Thermal conductivity
  - Volumetric heat capacity
  - Thermal linear expansion coefficient

These data were converted into the RETRAN input structure. Plots of the data, the analytical equations used to develop the data, and graphical and numerical comparisons were presented of the Framatome ANP data to the corresponding data in:

- the existing W fuel based model
- The International Nuclear Safety Center (INSC) Material Database, Argonne National Laboratory for the US Department of Energy
- NUREG/CR-6150 (MATPRO)

Generally, only minor differences in the data were observed. The most significant property differences are those associated with the M5 versus ZIRLO cladding.

Core Geometry Input

The Framatome ANP overlays were developed from Framatome ANP supplied dimensional data for the Framatome ANP fuel assemblies. All dimensional data were transmitted via documentation that was formally prepared and reviewed under Framatome ANP's 10 CFR 50 Appendix B QA program. Input changes were developed in the following areas:

- Core bypass geometry
  - Volume
  - Flow area
  - Flow diameter
- Active core geometry
  - Volume
  - Flow area
  - Flow diameter
- Reactor vessel flow path length and area
- Reactor vessel form loss coefficients
- Reactor core target pressure drops
- Active core inlet mass flow rate
- Geometry of the active core heat conductors

The calculation of each RETRAN input was documented in a reviewed engineering calculation and prepared in accordance with Dominion's 10 CFR 50 Appendix B Quality Assurance Program. The engineering calculation presents detailed comparisons of the Framatome ANP overlay parameters to the base model parameters in tabular format. The parameter changes represented minor adjustments with respect to the existing inputs.

Steady-state initializations were run with and without the Framatome ANP overlays to ensure adequate convergence of the new models. Detailed comparisons of the steady-state initialization results were presented in the engineering calculation in tabular format. Review of these results showed that there are only minor differences in the Westinghouse Fuel Based and Framatome ANP Fuel based models.

The modeling changes associated with Framatome ANP fuel fall within the restrictions and limitations of the Dominion core design and safety analysis codes.

Question 3:

Use of Framatome ANP fuel will require changes to various computer model inputs. Please discuss how the practices of NRC Generic Letter 83-11, Supplement 1, "Licensee Qualifications for Performing Safety Analyses", are applied in making these model changes.

Response:

General comment

The scope and applicability of GL 83-11 Supplement 1 is discussed in Attachment 1 to GL 83-11. An excerpt relevant to this discussion is as follows:

> "This attachment presents a simplified approach for qualifying licensees to use NRC-approved analysis methods. Typically, these methods are developed by fuel vendors, utilities, national laboratories, or organizations such as the Electric Power Research Institute, Incorporated (EPRI). To use these approved methods, the licensee would institute a program (e.g., training, procedures) that follows the guidelines below and notify the NRC that it has done so.
>
> The words 'code' and 'method' are used interchangeably within this document, i.e., a computer program. In many cases, however, an approved method may refer not only to a set of codes, an algorithm within a code, a means of analysis, a measurement technique, a statistical technique, etc., but also to selected input parameters which were specified in the methodology to ensure conservative results. In some cases, due to limitations or lack of appropriate data in the model, the code or method may be limited to certain applications. In these cases, the NRC safety evaluation report (SER) specifies the applicability of the methodology."

Dominion is proposing to apply the existing methodology of VEP-FRD-42 to the analysis of Framatome ANP fuel. Therefore GL 83-11, which involves code and methodology changes, is not directly applicable. However, the principles outlined in Attachment 1 to the GL have been followed in the development of Framatome ANP specific models (input changes) for use with existing, approved codes and methods. The process of Framatome ANP specific model development will be discussed in that context.

Dominion has established and uses a formal GL 83-11 program. Dominion notified the NRC of the establishment of this program in Reference 3.1. This program addresses all of the elements of GL 83-11, Supplement 1, Attachment 1 identified below:

- Application Procedures
- Training and Qualification of Licensee Personnel
- Comparison Calculations
- Quality Assurance and Change Control
- Error/Problem Reporting

Dominion's reload analysis methodology as set forth in VEP-FRD-42 has been developed and qualified in accordance with these principles. For example:

Application Procedures

Specific analytical steps for performing a reload analysis are outlined in the Nuclear Core Design (NCD) Manual and the Safety Analysis Manual (SAM). The NCD Manual is structured such that the calculational process is transparent to fuel type. Specific NCD code input varies according to fuel type as necessary (i.e., grid size differences, grid material difference, etc.). Detailed techniques for determining model input are provided in the NCD Manual and are supplemented by model setup calculations for previous fuel types, and by evaluation of proposed fuel changes in an operational impact assessment. The operational impact assessment is mandated by a departmental Implementing Procedure, which requires evaluations of proposed core changes in light of SOER 96-02.

The Safety Analysis Manual provides detailed calculational instructions for providing reload-specific thermal hydraulic evaluations as well as a chapter of guidance for the performance of analyses of the specific accidents presented in Chapters 14 and 15 of the Surry and North Anna UFSARs, respectively. Typically, accident reanalyses are not performed for core reloads, in that the key analysis parameters are found to be bounded by the assumptions in the accident analyses.

Quality Assurance/Change Control

Core Physics Models – The answer to Question 2 deals with the Framatome ANP changes of importance to the core design models. The changes were identified and evaluated in an operational impact assessment, and specific input changes were determined for Framatome ANP Lead Test Assembly (LTA) modeling using the same techniques used for other fuel types.

RETRAN Models - In preparation for application of the Dominion RETRAN model to Framatome ANP fuel, specific card (record) overlays to the RETRAN input cards were developed. These overlays were developed such that appending them to the end of the current, Westinghouse fuel based model creates a Framatome ANP-specific RETRAN model.

Specific changes modeled were discussed in detail in the Response to Question 2.

The Framatome ANP overlays were developed from the following data:

- Framatome ANP supplied fuel and clad properties data that are consistent with those used in the approved Framatome ANP safety analysis models. Formal documents developed under the Framatome QA program were developed to transmit this data.

- Framatome ANP supplied dimensional data for the Framatome ANP fuel assemblies. All dimensional data was transmitted via documentation that was formally prepared and reviewed under Framatome ANP's 10 CFR 50 Appendix B QA program.

Comparison Calculations

Previously submitted topical reports for PDQ Two Zone Models, NOMAD, and TIP/CECOR contain extensive model benchmarking information. In addition, the accuracy of power distribution predictions for Framatome ANP LTA fuel has been documented for three cycles of operation.

Dominion's RETRAN model has been benchmarked against the following items:

- Westinghouse analyses of record as published in the Surry and North Anna FSAR's in the 1970's and 1980's - see Section 5.2 of VEP-FRD-41A.
- Plant transient data, including:
  - Surry and North Anna pump coastdown tests - see Section 5.3 of VEP-FRD-41A
  - North Anna Unit 1's cooldown and safety injection transient September 25, 1979- See Section 5.3.3 of VEP-FRD-41A.
  - North Anna Unit 1's July 1987 Steam Generator Tube Rupture-see Section 3.2 of Attachment 1 to Letter 93-505, Supplemental Information on the RETRAN NSSS Model, August 10, 1993.
  - Westinghouse LOFTRAN calculations for the following:
    - Reactor trip with turbine trip
    - Turbine trip without direct reactor trip
    - Simultaneous loss of 3 reactor coolant pumps
    - See VEPCO Letter No. 376A, August 24, 1984.

These benchmark calculations have been studied and understood and support the conclusion that the Dominion RETRAN model provides a realistic representation of the Surry and North Anna reactor plants. Conservative results are ensured when the RETRAN model is used for licensing basis analyses through the use of appropriate input assumptions governing availability and performance of systems and components, core reactivity coefficients, and uncertainties in initial conditions.

Reference:

3.1 Virginia Power Letter to the NRC (Serial No. 00-087), dated March 15, 2000, Qualifications for Performing Safety Analyses, Generic Letter 83-11, Supplement 1.

Question 4:

The Dominion Topical Report on Reload Methodology (VEP-FRD-42, Revision 2) includes four computer codes or code modifications which have been implemented for use under the provisions of 10 CFR 50.59:

- PDQ Two Zone - replaced PDQ Discrete Model and the FLAME Model (Transmitted via Ref. 2 and 3 in VEP-FRD-42)
- NOMAD - was significantly modified (transmitted in Ref. 5 in VEP-FRD-42)
- TIP/CECOR - (Transmitted via Ref. 3 in VEP-FRD-42)
- RETRAN - code modifications (Transmitted via Ref. 7 in VEP-FRD-42)

References 2, 3 and 5 in VEP-FRD-42, Revision 2, and an additional letter not referenced in this topical (dated March 1, 1993) requested NRC review and approval of the associated topical reports for the first three codes listed. Dominion (VEPCO at the time) also recognized that these would need NRC approval because North Anna and Surry are COLR plants. For RETRAN, no review was requested, and the transmittal letter was for NRC information only. As such,

a. Have those topical reports/codes and code modifications been reviewed and approved for use by the NRC staff? If so, please provide a reference to the staff SERs. If not, then codes and models will need to be reviewed and approved to permit use in the COLR.

b. Have they been used by Dominion as part of the Reload Design Methodology? If so, why is their use acceptable and not a violation of the requirements for implementing a COLR? Generic Letter 88-16 requires that NRC approved methodology be referenced in the COLR, and VEP-FRD-42, Revision 1 is referenced in the COLR. VEP-FRD-42, Revision 1, and therefore the COLR does not reflect what Dominion is currently using as part of its Reload Methodology.

c. Please submit Technical Specification changes to incorporate references to actual methodology being used.

d. What procedures and controls do you use on the application of computer codes and models for core design and safety analysis? In other words, how does the core designer or safety analyst know he or she is using the right tools?

Response to 4a:

<u>PDQ Two-Zone Model</u>

The PDQ Two-Zone Model was transmitted via References 4.1 and 4.2:

Reference 4.1 requested approval of the 3-D coarse mesh PDQ model (the two-zone model) by the end of the 1st Quarter, 1991 to support the use of axially zoned flux suppression inserts (FSI's) in Surry Unit 1 Cycle 12.

Reference 4.2 reiterated the need for the 3D capability, to support FSI's, although first use had shifted to Cycle 13. We noted that to support the planned use of FSI's in Cycle 13 would require approval of the topical by the end of the 1st Quarter, 1993. Since the NRC review schedule would not support this, we proposed implementation of the methodology via 10 CFR 50.59 in advance of formal NRC approval of the reports. As noted in Reference 4.2, telephone conversations were held with the Staff on October 7 and 14, 1992 to discuss the 10 CFR 50.59 approach. Although the NRC could not concur with the specific application without formal review, the staff agreed with the use of 10 CFR 50.59 evaluations where applicable. Reference 4.2 documented these discussions. Dominion's request for formal review of the topicals was not withdrawn, although these changes were implemented via 10 CFR 50 .59.

On March 1, 1993 Dominion submitted Topical Report VEP-NAF-1, Supplement 1, entitled, "The PDQ Two-Zone Model," again for review and approval. The Supplement describes a coarse mesh 2-D model that is closely related to and used in conjunction with the 3-D model. We again stated our intent to implement the code via 10 CFR 50.59 prior to NRC review and approval, but requested concurrent review of the VEP-NAF-1 and Supplement 1.

The 10 CFR 50.59 approach to changing "elements of a methodology" as defined in NEI 96-07, Rev. 1 and endorsed by USNRC Regulatory Guide 1.187 is applicable in the case of the PDQ Two-Zone models. We refer specifically to NEI 96-07 Section 4.3.8, entitled, "Does the Activity Result in a Departure from a Method of Evaluation Described in the UFSAR Used in Establishing the Design Bases or in the Safety Analyses?"

The relevant discussion is as follows:

"... The following changes are not considered departures from a method of evaluation described in the UFSAR:

- Departures from methods of evaluation that are not described, outlined or summarized in the UFSAR (such changes may have been screened out as discussed in Section 4.2.1.3).

- Use of a new NRC-approved methodology (e.g., new or upgraded computer code) to reduce uncertainty, provide more precise results or other reason, provided such use is (a) based on sound engineering practice, (b) appropriate for the intended application and (c) within the limitations of the applicable SER. The basis for this determination should be documented in the licensee evaluation.

- Use of a methodology revision that is documented as providing results that are essentially the same as, or more conservative than, either the previous revision of the same methodology or another methodology previously accepted by NRC through issuance of an SER".

Subsection 4.3.8.1 of NEI 96-07 provides guidance for making changes to one or more elements of an existing method of evaluation used to establish the design bases or in the safety analyses. Specifically,

"4.3.8.1 Guidance for Changing One or More Elements of a Method of Evaluation

The definition of 'departure ...' provides licensees with the flexibility to make changes under 10 CFR 50.59 to methods of evaluation whose results are 'conservative' or that are not important with respect to the demonstrations of performance that the analyses provide. Changes to elements of analysis methods that yield conservative results, or results that are essentially the same, would not be departures from approved methods.

Conservative vs. Nonconservative Results

Gaining margin by changing one or more elements of a method of evaluation is considered to be a nonconservative change and thus a departure from a method of evaluation for purposes of 10 CFR 50.59. Such departures require prior NRC approval of the revised method. Analytical results obtained by changing any element of a method are 'conservative' relative to the previous results, if they are closer to design bases limits or safety analyses limits (e.g., applicable acceptance guidelines). For example, a change from 45 psig to 48 psig in the result of a containment peak pressure analysis (with design basis limit of 50 psig) using a revised method of evaluation would be considered a conservative change when applying this criterion. In other words, the revised method is more conservative if it predicts more severe conditions given the same set of inputs. This is because results closer to limiting values are considered conservative in the sense that the new analysis result provides less margin to applicable limits for making potential physical or procedure changes without a license amendment.

In contrast, if the use of a modified method of evaluation resulted in a change in calculated containment peak pressure from 45 psig to 40 psig, this would be a nonconservative change. That is because the change would result in more margin being available (to the design basis limit of 50 psig) for the licensee to make more significant changes to the physical facility or procedures.

Essentially the Same

Licensees may change one or more elements of a method of evaluation such that results move in the nonconservative direction without prior NRC approval, provided the revised result is 'essentially the same' as the previous result. Results are 'essentially the same' if they are within the margin of error for the type of analysis being performed. Variation in results due to routine analysis sensitivities or calculational differences (e.g., rounding errors and use of different computational platforms) would typically be within the analysis margin of error and; thus, considered 'essentially the same.' For example, when a method is applied using a different computational platform (mainframe vs. workstation), results of cases run on the two platforms differed by less than 1%, which is the margin of error for this type of calculation. Thus, the results are essentially the same, and do not constitute a departure from a method that requires prior NRC approval.

The determination of whether a new analysis result would be considered 'essentially the same' as the previous result can be made through benchmarking the revised method to the existing one, or may be apparent from the nature of the differences between the methods. When benchmarking a revised method to determine how it compares to the previous one, the analyses that are done must be for the same set of plant conditions to ensure that the results are comparable. Comparison of analysis methods should consider both the peak values and time behavior of results, and engineering judgment should be applied in determining whether two methods yield results that are essentially the same."

In the case of the PDQ Two-Zone models, the governing topical report documents extensive comparisons of these models to measured data and demonstrates that the Nuclear Reliability Factors (NRFs) documented in Topical Report VEP-FRD-45-A, "Nuclear Design Reliability Factors" remain bounding. Therefore, from a reload analysis perspective, the results with these new tools (elements of the VEP-FRD-42 methodology) are "essentially the same" and implementation via 10 CFR 50.59 is permissible.

<u>NOMAD</u>

Dominion uses the NOMAD 1-D core physics code to perform both reload design analyses and core operation evaluations. Use of this code and its associated model was approved by the NRC on March 4, 1985, with its issuance of Acceptance for Referencing of Licensing Topical Report VEP-NFE-1-A, "The VEPCO NOMAD Code and Model." As stated in VEP-NFE-1-A, verification of and improvements to the NOMAD code and model would continue to be made as more experience was gained in the application of the model to the units at the Surry and North Anna Power Stations. The primary reload safety analysis use of NOMAD is as one of the analytical tools (elements) of the Relaxed Power Distribution Control and Constant Axial Offset Control Methodologies. Use of NOMAD within the framework of those methodologies was not altered by the model update.

Letter 96-319 (Reference 4.4) documented the NOMAD code and model update. These changes were necessitated by the transition to 3-D PDQ (see discussion above). The NOMAD flux solution and axial nodalization were not altered. The updated NOMAD model was qualified against plant data and its fidelity to the data was found to be as good as or better than that of the original code and model. The Nuclear Reliability Factors currently applied in reload analyses were shown to remain appropriate and reload results obtained with the updated model are essentially the same as those obtained with the previous version. As such, the code and model updates do not constitute a change in the approved methodology of VEP-FRD-42 or the Code as described in VEP-NFE-1-A (see the discussion of NEI 96-07, Section 4.3.8, above).

TIP/CECOR

The CECOR code was reviewed and approved generically by the NRC and is documented in CENDP-153-P, Rev. 1-P-A. TIP-CECOR uses the same solution algorithm as CECOR, but is adapted to accept input from movable incore detectors as opposed to fixed detectors. Comparisons with experiments and development of uncertainties for TIP-CECOR are consistent with the CECOR topical report and with VEP-FRD-45-A, the Nuclear Design Reliability Factor topical report.

Additionally, comparisons between TIP/CECOR predictions and those from the previously approved INCORE code revealed that the two codes produce essentially the same results. Therefore, the adoption of TIP/CECOR as a replacement for INCORE represented a change to an element of the reload methodology that can be implemented via 10 CFR 50.59 under the guidance of NEI 96-07. Additionally, qualification of TIP/CECOR for Dominion use met the intent of the programmatic elements of Generic Letter 83-11, Supplement 1, Attachment 1.

RETRAN

Dominion's reload methodology incorporates the RETRAN-02 code. RETRAN-02 was generically approved by the NRC in a letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, "RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.

Dominion's RETRAN models and capability were approved by the NRC in a letter from C. O. Thomas (NRC) to W. L. Stewart, Acceptance for Referencing of Licensing Topical Report VEP-FRD-41, "Virginia Power Reactor System Transient Analyses Using the RETRAN Computer Code," April 11, 1985.

The RETRAN Topical SER recognized that model maintenance activities would be performed under the control of the utility 10 CFR 50 Appendix B QA program. The VEP-FRD-41 SER emphasized that the NRC viewed the primary objective of the report was to demonstrate Dominion's general capability for performing non-LOCA accident analyses:

- "The VEPCO topical report VEP-FRD-41, 'Reactor System Transient Analysis Using the RETRAN Computer Code,' was submitted to demonstrate the capability which VEPCO has developed for performing transient analysis using the RETRAN 01/M0D03 computer code."

- "The staff has reviewed the... VEPCO model descriptions and finds them acceptable for demonstrating understanding of the RETRAN code."
- "Based on the VEPCO RETRAN model and the qualification comparisons ..., the staff concludes that VEPCO has demonstrated their capability to analyze non-LOCA initiated transients and accidents using the RETRAN computer code."

Dominion has demonstrated that use of our models with RETRAN-02 versus RETRAN01 is an equivalent methodology. In a letter (Serial No. 85-753) dated November 19, 1985, Dominion showed that results with RETRAN-02 versus RETRAN-01 were essentially identical except for nonequilibrium pressurizer pressure behavior, where significant improvements were made in the RETRAN-02 solution scheme. This letter requested approval to use RETRAN-02 by February 1986 to support upcoming licensing applications; however, no formal NRC Staff review has been performed to date.

The VEP-FRD-41 SER further stated:

> "The staff requires that all future modifications of VEPCO RETRAN model and the error reporting and change control models should be placed under full quality assurance procedures."

Dominion followed these requirements in updating our RETRAN models. Updated models and the qualification results were documented consistent with our 10 CFR 50 Appendix B, QA program and provided to the NRC for information in letter (Serial No. 93-505) dated August 10, 1993.

It should be noted that the new model results were very similar to those obtained with the old models. No margins in key analysis results were gained. The new models have improved, more mechanistic Doppler reactivity feedback models and more detailed main steam system modeling. This resulted in some changes which were documented and well understood (see Letter 93-505).

While this model upgrade was not a code change, the qualification, documentation and implementation of these new models was done in a manner that meet the programmatic elements of Generic Letter 83-11, Supplement 1.

RETRAN models are code input, and represent an element of Dominion's RETRAN methodology as discussed in NEI 96-07. Because the results obtained with the new models met the "essentially the same" test, we believe that these model upgrades do not represent a change to a method of analysis as defined in 10 CFR 50.59 (c)(2)(viii).

Therefore, VEP-FRD-41A remains the applicable reference for Dominion's approved RETRAN capability.

Response to 4b:

Dominion has used these codes as part of its reload design methodology. However, with respect to the COLR, Dominion notes that the codes above are not listed in the COLR methods reference list in the Technical Specifications, because they do not represent analytical methods that determine core-operating limits. Dominion considers this treatment to be consistent with the guidance in Generic Letter 88-16, which discusses "methodology for determining cycle-specific parameter limits." PDQ and NOMAD represent tools that predict core performance and core parameter values, which are then compared to core operating limits. Similarly, TIP/CECOR processes core surveillance data to confirm that core parameters are behaving as predicted by PDQ and NOMAD and that the operating limits are continuously met. RETRAN provides transient system thermal hydraulic responses that are used in conjunction with the COBRA and LYNXT codes to perform transient DNB calculations for Chapter 15 accidents. The Nuclear Enthalpy Rise Hot Channel Factor (FΔH) limit in the COLR is established using COBRA and LYNXT in conjunction with the Reactor Core Safety Limits, and not by RETRAN. Similarly the total peaking factor limit (FQ) in the COLR is established by the referenced, approved LOCA methodology, not by the neutronics codes.

Although VEP-FRD-42, Rev. 1 was not formally revised to reflect changes to these codes and models, it was updated via supplements sent with references 4.3 and 4.4. In neither case was there any NRC request or directive given to revise the topical to incorporate these changes. In particular, Reference 4.3 summarizes several changes relevant to VEP-FRD-42, Rev. 1-A and states:

> "These changes have effectively superseded portions of VEP-FRD-42, Rev. 1-A. Supplement 1 to VEP-FRD-42, Rev. 1-A (enclosed) consolidates and summarizes these changes for your information."

Dominion therefore, considers that these supplements are part of VEP-FRD-42, Rev. 1 and that VEP-FRD-42, Rev. 1 continues to represent Dominion's reload methodology for Westinghouse fuel. It is not Dominion's intention to change our reload methodology as outlined in VEP-FRD-42, Rev. 2 under the provisions of 10 CFR 50.59. However, there are analytical tools, which form elements of the methodology, which can be changed under the provisions of 10 CFR 50.59(c)(2)(viii) as discussed in NEI 96-07 Section 4.3.8.

It is Dominion's intent to apply this guidance of NEI 96-07, Rev. 1, as endorsed by Regulatory Guide 1.187, in determining the applicability of 10 CFR 50.59 to proposed changes to analytical tools which support our reload methodology. The qualification and benchmarking of new elements of the methodology for making this determination will be performed and documented in accordance with the provisions of our quality assurance program.

Response 4c:

The code/model updates discussed in the response to 4a and 4b, above, have been incorporated into VEP-FRD-42, Rev. 2 by referencing the appropriate documentation. Since VEP-FRD-42 is currently referenced in the Technical Specifications no additional changes are necessary.

Response 4d:

A. Production Codes

Core designers and safety analysts have access to a controlled Production Code List.

The Production Code List includes the code version, the effective date, a reference to the applicable code file (which contains the software development, qualification and release documentation), the Code Manager and applicable references documenting the qualification and implementation of the code. This documentation is prepared and peer reviewed in accordance with applicable quality assurance procedures. (The Code Manager is an individual designated by the Department Manager to ensure the required code documentation is completed for new codes and changes to existing codes).

Engineers refer to the List when referencing the name and version of a computer code used to perform design calculations. This procedure ensures that any computer code referenced in a Calculation is available for production work and that the appropriate version of the code is used.

The code version and release date is printed on the output header of all computer calculations. Computer code versions are required to be included as formal references in the engineering calculations which document production applications (e.g., reload calculations).

Dominion software control procedures require that qualified code users be notified when modifications to a code are made.

B. Models

A procedure governs the development and control of Nuclear Analysis and Fuel models. A model is defined as a standardized, controlled set of plant specific input to a computer code. The physical model consists of one or more electronic input files. Models are treated as controlled documents.

Production model input files are write-protected with only authorized personnel given change authority, or monitored in such a way that the Model Manager can determine whether the files have been modified. Model users are responsible for ensuring that the appropriate model is used correctly in an analysis.

Recent changes to applicable production codes and models are discussed as part of the reload design initialization process (see VEP-FRD-42, Rev. 2 Section 3.2.1).

References:

4.1 Letter from W. L. Stewart (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "Virginia Electric and Power Company, Surry Power Station Units 1 & 2, North Anna Power Station Units 1 and 2 Topical Report–PDQ Two Zone Model," Serial No. 90-562, October 1, 1990.

4.2 Letter from W. L. Stewart (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "Virginia Electric and Power Company, Surry Power Station Units 1 & 2, North Anna Power Station Units 1 and 2 Topical Report Use Pursuant to 10 CFR 50.59," Serial No. 92-713, November 25, 1992.

4.3 Letter from M. L. Bowling (Virginia Electric and Power Company) to U. S. Nuclear Regulatory Commission, "Virginia Electric and Power Company, Surry Power Station Units 1 & 2, North Anna Power Station Units 1 and 2, Supplement 1 to VEP-FRD-42 Revision 1-A, Reload Nuclear Design Methodology Modifications," Serial No. 93-723, December 3, 1993.

4.4 Letter from S. P. Sarver (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "Virginia Electric and Power Company, North Anna Power Station Units 1 & 2, Surry Power Station Units 1 and 2, Supplemental Information for the NOMAD Code and Model, Reload Nuclear Design Methodology, and Relaxed Power Distribution Control Methodology Topical Reports," Serial No. 96-319, November 13, 1996.

Question 5:

VEP-FRD-42, Revision 1 included the code or model used to calculate each of the Key Analysis Parameters within the sections of the report, which discussed each parameter. This is not done in Revision 2. Please provide a listing of the code or model used to calculate each Key Analysis Parameter used in the reload analysis methodology. Does the use of Framatome ANP fuel introduce any new Key Analysis Parameters?

Response:

The models currently used to calculate each parameter are provided below, in terms of the key parameter list from Table 2 of VEP-FRD-42, Revision 2. It was determined that the Framatome ANP fuel required the addition of one key parameter (item 28 below). This parameter, maximum linear heat generation rate versus burnup, is used in the NRC-approved Framatome ANP methodology for cladding stress evaluations. The code or model currently used to calculate each parameter is listed in the following table. The name PDQ refers to the PDQ two-zone 3D model.

KEY ANALYSIS PARAMETER	CODE OR MODEL
1) Core Thermal Limits (F)	COBRA/LYNXT
2) Moderator Temperature (Density) Coefficient (NS)	PDQ
3) Doppler Temperature Coefficient (NS)	PDQ
4) Doppler Power Coefficient (NS)	PDQ
5) Delayed Neutron Fraction (NS)	PDQ
6) Prompt Neutron Lifetime (NS)	NULIF
7) Boron Worth (NS)	PDQ
8) Control Bank Worth (NS)	PDQ/NOMAD
9) Rod Worth Available for Withdrawal (S)	PDQ/NOMAD
10) Ejected Rod Worth (S)	PDQ/NOMAD
11) Shutdown Margin (NS)	PDQ/NOMAD
12) Boron Concentration for Required Shutdown Margin (NS)	PDQ
13) Reactivity Insertion Rate due to Rod Withdrawal (S)	PDQ/NOMAD
14) Trip Reactivity Shape and Magnitude (NS)	PDQ/NOMAD
15) Power Peaking Factors (S)	PDQ/NOMAD
16) Maximum F_Q * P (S)	PDQ/NOMAD
17) Radial Peaking Factor (S)	PDQ
18) Ejected Rod Hot Channel Factor (S)	PDQ/NOMAD
19) Initial Fuel Temperature (F)	PAD /TACO3
20) Initial Hot Spot Fuel Temperature (F)	PAD /TACO3
21) Fuel Power Census (NS)	PDQ/NOMAD
22) Densification Power Spike (F)	PAD /TACO3
23) Axial Fuel Rod Shrinkage (F)	PAD /TACO3
24) Fuel Rod Internal Gas Pressure (F)	PAD /TACO3
25) Fuel Stored Energy (F)	PAD /TACO3
26) Decay Heat (F)	ANSI ANS-1979 ANSI ANS-1971
27) Maximum Linear Heat Generation Rate (LHGR) (S)	PDQ/NOMAD
28) Maximum LHGR Vs. Burnup (F)	PDQ/NOMAD

Parameter Designation
S: Specific
NS: Non-specific
F: Fuel Performance and Thermal-Hydraulics Related

Question 6:

Regarding Section 2.2.2.1 - Reactivity Coefficients and Defects:

a. Revision 1 discussed a set of four calculations performed to determine temperature and power coefficients at HZP, and an additional four cases to determine the coefficients at power. The Revision 2 methodology includes two cases at ±5°F or ±10°F about the nominal temperature for the temperature coefficients, and two cases at ±5% or ±10% about the nominal power for the power coefficients. Please provide the technical basis supporting this change in methodology.
b. The cases at ±10°F or ±10% were not included in Revision 1 methodology. Please provide the technical basis for these cases.
c. Please discuss the procedures or processes by which the Dominion analyst determines whether to use ±5 or ±10.

Response:

Parts a and b:

Two cases are used for each coefficient. Four cases are still required to determine all three coefficients (ITC, DTC, and MTC). The discussion of HZP coefficients simply reflects the calculation of individual coefficients because all three coefficients are not required at all conditions.

The choice of ±5°F or ±10°F does not have a significant effect on most coefficients (particularly the DTC) because they behave nearly linearly versus temperature over this small a temperature range. Mathematically, as long as the defect is no more complex than a quadratic function of temperature, there is no effect at all in the choice of temperature difference, provided that a centered difference is used. In general, ±5°F is used for all but the DTC. The DTC is always small in magnitude and, therefore, is more susceptible to K-effective convergence tolerance. A range of ±10°F reduces the influence of convergence tolerance. The defining methodology features in the calculation of coefficients are:

1) changing only the variable(s) of interest (fuel temperature, moderator temperature or both, or core power), and
2) the use of a centered difference about the desired point over a range large enough to get a significant change but small enough that the answer still represents the derivative.

As indicated, valid technical reasons may arise which lead to a change in the exact choice of temperature difference or the specific input used to calculate a coefficient. The above discussion also applies to the at-power ITC, DTC, and MTC cases. As in the case of the temperature coefficients, the use of ±10% power for power coefficients does not represent a significant change due to the nearly linear nature of the power coefficients versus power. The primary reason for using ±10% is to minimize 3D-model

THF convergence tolerance on the coefficients. We do not view these specific input changes as changes to the reload methodology.

Part c:

The analyst uses standard techniques described in the core design procedures. These techniques, including the choice of temperature or power change are not changed unless a valid new technical reason arises. A change to the standard technique requires peer review and management approval.

Question 7:

Section 2.3 - Analytical Model and Method Approval Process was added in Revision 2 and discusses the acceptable means by which either analytical models or methods can achieve approved status for use in reload methodology. The first method listed allows reload methodology changes to be implemented in accordance with the provisions of 10 CFR 50.59. The NRC staff does not accept this option as a means to change reload methodology. Implementation under 10 CFR 50.59 would require that new or different methods have already been reviewed and approved by the NRC for the intended application.

Response:

Dominion did not and does not change the reload methodology as outlined in VEP-FRD-42, Rev. 2 under the provisions of 10 CFR 50.59. However, there are analytical tools, which form elements of the methodology, which can be and have been changed under the provisions of 10 CFR 50.59(c)(2)(viii) as discussed in NEI 96-07, Section 4.3.8 (see our response to Question 4, above for further discussion).

The qualification and benchmarking of new or revised inputs or elements of the methodology are performed and documented in accordance with the provisions of our quality assurance program. Dominion then applies the guidance of NEI 96-07, Rev. 1, as endorsed by Regulatory Guide 1.187, in determining the applicability of 10 CFR 50.59 to the proposed changes.

This practice is analogous to that used for previous model updates prior to the issuance of NEI 96-07. For example, application of the 50.59 process to the PDQ model changes (and later the NOMAD and TIP/CECOR changes) was focused on the key issues of whether the change created an unreviewed safety question (USQ), maintaining the "margin of safety," and whether the change involved a change to a Technical Specification. The SER for prior model approvals were reviewed to ascertain the NRC basis for previous approval. In particular, the PDQ Two Zone model was found to be an equivalent replacement of the previous models used for the same purposes inside the existing reload methodology framework and hence the change was determined not to be a USQ. The validation process was at least as broad as for the earlier models, with far more available data. Although the data supported reductions in some uncertainty factors, the existing uncertainty factors were maintained (no reduction in margin of safety). The process used is functionally equivalent to changing elements of the method under the current 50.59 process. This was an internal review process using the same criteria as the original review as described in associated NRC SERs and using appropriate screening techniques under 50.59. Finally, since PDQ was not directly referenced in the COLR, implementation of the model upgrades did not require a change to the Technical Specifications. As discussed in the response to Question 4b, PDQ is not listed among the analytical methods supporting the COLR in Technical Specifications since it is not used to determine values for core operating limits.

The process for qualifying the new RETRAN models was analogous. The qualification tests performed included comparisons between the new and old models as well as to plant transient data. The qualification supported the conclusion that the new models were an equivalent replacement of the transient analysis element of Dominion's reload methodology.

Question 8:

Regarding Section 3.3.2 - Safety Analysis Philosophy, please discuss the procedural or process type of guidance available to the Dominion analyst for determining whether to evaluate or reanalyze a particular transient. This would be important if a key reload parameter value exceeds the current limit in the reference safety analysis, or if the parameter impact is difficult to quantify.

Response:

Quantitative evaluation of a small departure from a parameter limit of parameter limits may be made in one of several ways. First, if the interplay between the various key safety parameters in determining accident response is well defined, margin in one parameter may be used to offset a small departure in another parameter. A second method of quantitative evaluation involves using tradeoffs of known sensitivities. This process is best defined by presenting some examples:

- Studies performed by Dominion and others have shown that a key parameter in determining the severity of the core power response to a rod ejection event is the ejected rod worth in units of dollars (delta k/k ejected rod worth/delayed neutron fraction). For the case of a cycle-specific departure from the minimum delayed neutron fraction, the safety analyst can take advantage of available cycle-specific margin in ejected rod worth by showing that the ejected worth in dollars is less than the worth assumed in the safety analysis.

- For some reload cycles where small departures (a few percent) from an accident specific limit occur, these studies can be used to show that margin in another key parameter that influences the same accident offsets the departure. For example, the end of cycle (EOC) least negative moderator temperature coefficient is a key safety parameter for the rod ejection accident, although its influence is relatively weak. For one recent cycle, a small departure from the limit for this parameter was shown to be offset by large margins in the calculated ejected rod worth, which strongly influences the accident analysis results. These sensitivities are documented in VEP-NFE-2-A.

The general philosophy followed in performing an accident evaluation as opposed to a reanalysis is that the analyst must be able to clearly demonstrate that the results of an analysis performed with cycle-specific input would be less severe than the results of the reference analysis. In other words, in performing the evaluation, no credit is taken for margin between the reference analysis results and the design basis criteria, even though this margin may be substantial. In some cases the analyst and/or reviewer may determine that a cycle specific transient analysis should be performed to verify that the reference analysis remains bounding. No specific quantitative criteria have been established for making this determination, but every instance in which an evaluation (as opposed to a reanalysis) of a key parameter departure is performed must be documented. In the documentation the analyst presents the exact numerical values pertaining to the departure from a limit and a detailed discussion of the reasoning and approach used in reaching a conclusion regarding the parameter in question. This documentation is subject to peer review and approval. The results of these cycle specific evaluations are summarized in the Reload Safety Evaluation (RSE) report.

Question 9:

In Section 3.3.2 - Safety Analysis Philosophy, it is stated that, "The methods that will be employed by Dominion to determine these key parameters will be consistent with the methods documented in References 9, 12, and 14" [of VEP-FRD-42, Revision 2]. References 12 and 14 are Westinghouse WCAP methodologies for reload safety evaluations, and power distribution control and load following procedures. Please discuss the evaluations performed to verify that these methodologies are also applicable for Framatome ANP fuel.

Response:

This section of VEP-FRD-42, Revision 2 defines 3 types of key parameters used to characterize the behavior of reload cores to various postulated accidents. The detailed calculation of specific key parameter values for a reload core is performed using the applicable core design or fuel design tools, dependent upon the parameter involved. The reload safety analysis framework involves evaluating the key parameter values determined for each reload to verify that margin exists between the reload value and the limiting value assumed in the reference safety analysis. This bounding value approach requires the existence of certain predefined relationships that identify the relevant key parameters for a given postulated accident, and their sensitivities (i.e., direction of most limiting effect).

References 9 and 14 of VEP-FRD-42, Revision 2 describe the detailed methodology for defining achievable core power distributions and associated operating limits for two different control schemes employed in Dominion analyses. Reference 9 defines the Dominion-developed Relaxed Power Distribution Control (RPDC) methodology and Reference 14 defines the Westinghouse-developed Constant Axial Offset Control (CAOC) methodology. Each of these methodologies involves the simulation, using detailed nuclear core design codes and models, of a defined number of perturbed core states and the corresponding power distributions. Each of these methodologies is used to determine the limits of normal core operation that will ensure that localized core power distributions remain within the values assumed as initial conditions in the accident analyses. Both methodologies are dependent upon defining proper design input details that characterize the core neutronic behavior. The required design input items involve detailed inputs such as nuclear cross-sections, geometry (fuel pellet, fuel rod and fuel assembly) and enrichment and reactor system inputs such as power, temperature and flowrate. There are several features of the Framatome ANP fuel that differ from the existing fuel design, including: theoretical density, use of Mid-Span Mixing Grids and use of alloy M5. The evaluation of these changes has concluded that each represents alteration of a detailed design input, but not a change that affects the reload methodology. Each of these features of the Advanced Mark-BW fuel was reviewed and found to be within the existing capability and range of applicability of the nuclear core design and safety analysis tools. It was thus concluded that the existing methodologies documented in References 9 and 14 could be used for analysis of the Advanced Mark-BW fuel with its slightly different features.

Reference 12 of VEP-FRD-42, Revision 2 documents the Westinghouse-developed reload evaluation methodology that supports the generic basis for the Dominion reload methodology. The Westinghouse methodology defines specific key parameters for use in accident analyses and their limiting directions for consideration in reload evaluations. Reference 12 is referenced in this sense, in that it defines part of the overall framework that constitutes the Dominion methodology. The changes associated with an alternate fuel design may be of two types: 1) changes that reflect physical fuel design features and 2) changes that reflect licensed analysis approaches or requirements. The Advanced Mark-BW fuel design was assessed for both types of change with respect to applicability of the Reference 12 methodology. It was concluded that none of the physical design features invalidate the key parameter definitions or usage as cited in Reference 12 and VEP-FRD-42, Revision 1. The review associated with potential licensed analysis approaches determined that the Framatome ANP fuel required an additional key parameter, which is reflected in Table 2 of VEP-FRD-42, Revision 2. This parameter, maximum linear heat generation rate versus burnup, is used in the NRC-approved Framatome ANP methodology for cladding stress evaluations. This parameter can be calculated with existing nuclear design codes. This review has demonstrated that the citation of Reference 12 as used within the reload methodology of VEP-FRD-42, Revision 2 is valid for reload evaluation of the Framatome ANP fuel.

Question 10:

Please identify and provide a reference for the fuel lattice physics code used to calculate the prompt neutron lifetime key analysis parameter (Section 3.3.3.5). Include a reference to the NRC staff SER approving this code. Please verify and provide the technical basis for the application of this code to expected fuel designs.

Response:

The lattice code referred to in Section 3.3.3.5 is NULIF, which is the same code used in VEP-FRD-42, Rev. 1. NULIF was originally reviewed as part of VEP-FRD-19A (Ref. 10.1) and the prompt neutron lifetime reliability factor was approved in VEP-FRD-45A (Ref. 10.2). NULIF is a pin cell neutron spectrum / isotopic depletion code. The input to NULIF (i.e., fuel density, fuel enrichment, clad material, fuel pin geometry, soluble boron concentration, depletion power, depletion interval, etc.) for Framatome ANP fuel is not significantly different than for Westinghouse fuel. NULIF is used for both Surry (15x15 lattice) and North Anna (17x17 lattice), and the differences between 15x15 and 17x17 fuel are more significant than the differences between Framatome ANP and Westinghouse fuel.

Reference:

10.1 M. L. Smith, "The PDQ07 Discrete Model," VEP-FRD-19A (July 1981).

10.2 Letter from United States Nuclear Regulatory Commission to Mr. W. N. Thomas, Virginia Electric and Power Company, "Acceptance for Referencing of Topical Report VEP-FRD-45 'Nuclear Design Reliability Factors,' " August 5, 1982.

Question 11:

The dropped RCCA(s) event (dropped rod or dropped bank) is evaluated using the methodology described in Westinghouse WCAP-11394-P-A (Reference 15 of this topical report). Please discuss the evaluation performed to verify that this methodology is also applicable for Framatome ANP fuel.

Response:

The dropped rod methodology of WCAP-11394 requires that three analyses be performed in order to perform an evaluation of the dropped rod event. These analyses, referred to as transient, nuclear, and thermal-hydraulic analyses, provide (1) the statepoints (reactor power, temperature, and pressure), (2) the radial power peaking factor, and (3) the DNB analysis at the conditions determined by items 1 and 2, respectively. These analyses are performed using a parametric approach so that cycle specific conditions may be evaluated using the data generated in the three analyses mentioned above.

Westinghouse, in WCAP-12282 (Reference 11.1), provided generic guidelines that established a common approach for implementation of the revised dropped rod methodology. WCAP-12282 indicated that the core physics correlations and transient statepoints generated for the methodology described in WCAP-11394 apply to all Westinghouse plants with 12 or 14 foot cores. However, due to the plant specific nature of the core physics characteristics and the thermal-hydraulic dropped rod limit lines, a generic safety analysis which bounds all plants is not feasible. Therefore, for every fuel cycle, plant specific data are combined with the appropriate set of correlations and statepoints to verify that the DNB design basis is met for the dropped rod event. The transient statepoints have been generated to be independent of reload considerations. The thermal-hydraulic limit lines are determined on a plant specific basis using currently licensed thermal-hydraulic models. The core physics data required for the analysis are generated during the normal course of the reload design.

The NRC, in Question No. 7 of the request for additional information for WCAP-11394, queried whether the plant/cycle specific calculations are really performed for the items mentioned, or have bounding values been used. The response in WCAP-11394-P-A states that "...the statepoints and R factors are not required to be calculated on a plant or cycle specific basis. Figures IV-1 through IV-8 show the generic applicability of the models used for various fuel types and cycle designs. However, the statepoints and/or R factors would be reassessed for new plants or fuel designs."

As described in WCAP-11394, the transient analysis consists of generating statepoint information (reactor power, temperature, and pressure) for a large number of dropped rod transient events. These statepoints cover a range of reactivity insertion mechanisms for use in the nuclear analysis: the worth of the dropped rod, the moderator temperature coefficient, and the total rod worth available in the control bank which is withdrawn by the Rod Control System when it attempts to restore power to the nominal value. Statepoint data for a large number of transient events, generated by Westinghouse, were used in application of this methodology to North Anna and Surry Power Stations. The statepoint data are influenced by NSSS and protection system features, and were generated to accommodate a wide range of potential core physics conditions. The validity of the statepoint data is, thus, not affected by the transition to Framatome ANP fuel.

The dropped rod methodology employs a bounding empirical correlation between dropped rod worth, $F\Delta H$, and MTC to relate the power change associated with a dropped rod (or rods) to the increase in peaking factor caused by the dropped rod. In order for this correlation to become non-conservative, either the peaking factor change associated with a dropped rod of a particular worth must increase or the power change associated with the dropped rod reactivity insertion must decrease. As indicated in the response to Question 2, the core physics characteristics of the Framatome ANP fuel are nearly identical to the Westinghouse fuel it will replace. There is no change in loading pattern strategy associated with Framatome ANP fuel that would cause a change in the range of dropped rod worth or in the relationship between dropped rod worth and peaking factor increase. Reload cores, therefore, will not respond in a fundamentally different way to the dropped rod event due to the use of Framatome ANP fuel.

The final portion of the dropped rod methodology is the DNB analysis at the conditions determined from the statepoints (reactor power, temperature, and pressure) and the radial power peaking factor. For the DNB analysis, the methodology employs dropped rod limit lines that are representations of the core conditions (inlet temperature, pressure, core power level, and $F\Delta H$) for which the DNBR is equal to the DNBR design limit. The dropped rod limit lines for the resident Westinghouse fuel were shown to be applicable for both fuel types.

Therefore, the methodology described in Westinghouse WCAP-11394-P-A is applicable for Framatome ANP fuel.

Reference:

11.1 R. L. Haessler, "Implementation Guidelines for WCAP-11394 (Methodology for the Analysis of the Dropped Rod Event)," WCAP-12282, June 1989

Question 12:

Section 3.5 - Nuclear Design Report, Operator Curves, and Core Follow Data included the following changes to the list of design report reload parameters:

a. Iodine has replaced Samarium worth, and
b. K-effective at refueling conditions as a function of temperature and rod configuration has been removed from the list.

Please provide the technical basis for these changes.

Response:

Part a:

Iodine has not replaced samarium. Iodine has been added to the xenon information. Samarium has been replaced by "Reactivity due to isotopic decay," which includes the contribution of samarium as well as less significant nuclides which build up or decay after shutdown on a time scale similar to samarium.

Part b:

The K-effective for refueling data is now transmitted to the power station prior to issuance of the design report. This was an administrative change to support outage planning and not a change in methodology.